5.4 FUEL STORAGE

APPLICABILITY

Applies to the capacity and storage arrays of new and spent fuel.

OBJECTIVE

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

SPECIFICATION

a. Criticality

1. The spent fuel storage racks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum enrichment of ~~52.3~~ grams Uranium-235 per axial centimeter;

   b. k_{eff} < 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties.

2. The new fuel storage racks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum enrichment of ~~49.2~~ grams Uranium-235 per axial centimeter;

   b. k_{eff} < 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties;

   c. k_{eff} < 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties.

3. The spent fuel pool is filled with borated water at a concentration to match that used in the reactor refueling cavity and refueling canal during REFUELING OPERATIONS or whenever there is fuel in the pool.

b. Capacity

The spent fuel storage pool is designed with a storage capacity of 990 assemblies and shall be limited to no more than ~~982~~ fuel assemblies.

ATTACHMENT 3

Letter from C. R. Steinhardt (WPSC)

to

Document Control Desk (NRC)

Dated

May 7, 1998

Proposed Amendment 154

Affected TS Page:

TS 5.4-1

5.4 FUEL STORAGE

APPLICABILITY

Applies to the capacity and storage arrays of new and spent fuel.

OBJECTIVE

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

SPECIFICATION

a. Criticality

1. The spent fuel storage racks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter;

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties.

2. The new fuel storage racks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter;

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties;

   c. $k_{eff} < 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties.

3. The spent fuel pool is filled with borated water at a concentration to match that used in the reactor refueling cavity and refueling canal during REFUELING OPERATIONS or whenever there is fuel in the pool.

b. Capacity

The spent fuel storage pool is designed with a storage capacity of 990 assemblies and shall be limited to no more than 990 fuel assemblies.

ATTACHMENT 4

Letter from C. R. Steinhardt (WPSC)

to

Document Control Desk (NRC)

Dated

May 7, 1998

Proposed Amendment 154

Radiological Assessment of
Increased Mass of Uranium-235
per Axial Centimeter

A radiological safety analysis was previously performed (Reference 2) for design enrichments up to 5.00 wt% uranium 235(U^{235}) and for assembly average burnups up to 60 GWD/MTU. That analysis was performed using an initial heavy metal loading (IHM) of 379.06 kgU (which corresponds to 52.3 grams of U^{235} per axial centimeter) and evaluated

(1) adequacy of shielding,
(2) fuel handling accident doses,
(3) adequacy of spent fuel pool ventilation system, and
(4) adequacy of spent fuel pool cleanup system.

These items were evaluated using conservative source terms calculated by the ORIGEN2 code. The analysis in Reference 5 demonstrates that the conclusions of the above items remain valid and applicable for the new Siemens Power Corporation 14 x 14 heavy fuel assembly with an IHM of 406.74 kgU ± 1%. The proposed limit, 56.067 grams U^{235} per axial centimeter, corresponds to the maximum IHM with the 1% tolerance applied, 411kgU. The conclusions of Reference 5 are restated and summarized below.

Adequacy of Shielding

The use of fuel with an initial enrichment of 5.00 wt% U^{235}, an initial IHM of 406.74 kgU ± 1%, and burnups up to 60 GWD/MTU will not significantly increase any dose rates where shielding is required. Table 1 compares the photon emission rates for an assembly with IHM of 379.06 and 411 kgU ± 1% at 100 hours after shutdown. One hundred hours is the technical specification limit for the minimum time that must elapse after subcriticality before irradiated fuel can be moved. The comparison shows that increasing the IHM has a negligible effect on the total energy emission and shielding therefore continues to be adaquate.

Fuel Handling Accident In Containment (FHAIC)

Kewaunee Technical Specification Amendment 132 permitted operation with the containment airlock doors open during fuel handling activities. On April 7, 1998, WPSC provided an evaluation of the fuel handling accident in containment to support continued NRC review of Amendment 132 (Reference 6). For expediency, the FHAIC analysis was performed with IHM consistent with that proposed for this amendment. Based upon the analysis, WPSC concluded that radiological consequences of a fuel handling accident in containment were acceptable. This analysis is still under NRC review.

Fuel Handling Accident Outside Containment (FHAOC)

The analysis for a FHAIC assumes release of the limiting fuel assembly gap activity without credit for holdup or filtration. Kewaunee Technical Specifications require periodic surveillance and operation of the Spent Fuel Pool Sweep and Exhaust System which includes charcoal filtration during refueling operations. This ventilation system supplies air across the spent fuel pool to be exhausted through the charcoal filters. With this additional protection, WPSC has concluded that any postulated release and consequential dose from a fuel handling accident outside containment are bounded by the results of the FHAIC.

Adequacy of Spent Fuel Pool Ventilation System

The adequacy of the spent fuel pool ventilation system was previously evaluated in Reference 2. The evaluation was based on the current capacity of 990 spent fuel assemblies, a maximum enrichment of 5.0 wt% U^{235}, a maximum heavy metal loading 379 kgU/assembly, and a maximum burnup of 60 GWD/MTU. For the same reasons discussed in Reference 2, the increase in the imitial heavy metal loading from 379 kgU to 411 kgU does not prevent the spent fuel pool ventilation system from performing its intended function.

Table 2 presents the relative nuclide activity differences between an assembly with a 379.06 kgU loading and the new Siemens 14 x 14 heavy fuel assembly with a 411 kgU loading. The single assembly activities were calculated using a conservative power history via the ORIGEN2 code. Tables 1 and 2 show that increasing the IHM has a small effect on the energy release and activities of the volatile nuclides for a single assembly.

Based on the conclusions of the Reference 2 analysis and the comparisons in Tables 1 and 2, the use of fuel with an initial enrichment of 5.00 wt% U^{235}, an initial IHM of up to 411 kgU, and burnups up to 60 GWD/MTU will not have a significant effect upon the spent fuel ventilation system. The spent fuel ventilation system will therefore remain adequate for use with the new Siemens Power Corporation heavy assemblies.

Adequacy of Spent Fuel Pool Cleanup System

The adequacy of the spent fuel pool cleanup system was previously evaluated in Reference 2. The evaluation was based on the current capacity of 990 spent fuel assemblies, a maximum enrichment of 5.0 wt% U^{235}, a maximum heavy metal loading of 379 kgU/assembly, and a maximum burnup of 60 GWD/MTU. For the same reasons discussed in Reference 6, the increase in the initial heavy metal loading from 379 kgU to 411 kgU does not prevent the spent fuel pool cleanup system from performing its intended function.

Other Design Transients and Accidents

WPSC performed a review of the design transients and accidents to assess the impact of the increased IHM on the radiological consequences of these events. Increased IHM results in a slight increase in the fuel assembly total activity. Of these design basis transients and accidents, two were identified as being impacted by the proposed change: Fuel damage from a turbine missile and the loss of coolant accident. Each is discussed below.

Turbine Missile Damage to Spent Fuel

Table 2 presents the relative nuclide activity differences between an assembly with a 379.06 kgU loading and the new Siemens 14 x 14 heavy fuel assembly with a 411 kgU loading. Since this difference is negligible, WPSC has concluded that any postulated release and consequential dose from a turbine missile accident are not significantly changed from the previous analysis.

Large Break Loss of Coolant Accident (LB-LOCA)

The Updated Safety Analysis Report discusses the postulated doses from the design basis accident. As in NRC Reg. Guide 1.4, it is assumed that 100 percent of the noble gases and 50 percent of the iodines in the core's fission product inventory will be released to the reactor containment vessel.

Analyses have shown that the total fuel assembly activity is only slightly sensitive to IHM. An increase in activity of less than 10% is expected. The current design basis analysis results for two-hour thyroid and whole body doses are 13% and 8% of 10 CFR 100 limits respectively. WPSC has therefore concluded that any postulated release and consequential dose from an LB-LOCA are not significantly changed from the previous analysis and remain well below 10 CFR 100 limits..

WPSC therefore concludes that the proposed changes do not significantly increase the radiological consequences of previously evaluated accidents.

Table 1. Comparison of photon energy emission rates for a single assembly with different IHM loadings and an enrichment of 5.00 wt% U^{235} as a fnnction of burnup - at 100 hours after shutdown.

		(Based on a radial power peaking factor of 1.46)					
ORIGEN2 Group		Ratio=(IHM 411 kgU/IHM 379.06 kgU) for several burnups (GWD/MTU)					
	Emean	14	26	36.5	46.5	48	60
1	$1.50x10^{-2}$	0.99195	0.97934	0.96989	0.96356	0.96150	0.95784
2	$2.50x10^{-2}$	0.99910	0.99728	0.99631	0.99547	0.99549	0.99545
3	$3.75x10^{-2}$	1.00060	1.00018	0.99945	0.99747	0.99717	0.99501
4	$5.75x10^{-2}$	0.98616	0.96569	0.95900	0.95556	0.95496	0.96077
5	$8.50x10^{-2}$	0.99562	0.98793	0.98080	0.97499	0.97352	0.97025
6	$1.25x10^{-1}$	0.99559	0.98792	0.98135	0.97552	0.97450	0.96979
7	$2.25x10^{-1}$	0.99450	0.98499	0.97830	0.97193	0.97050	0.96692
8	$3.75x10^{-1}$	0.99826	0.99583	0.99397	0.99177	0.99151	0.98942
9	$5.75x10^{-1}$	0.99672	0.99285	0.99023	0.98739	0.98693	0.98545
10	$8.50x10^{-1}$	1.00139	1.00219	1.00207	1.00079	1.00078	0.99542
11	1.25	0.98711	0.97400	0.95920	0.94421	0.94153	0.93055
12	1.75	1.00166	1.00273	1.00279	1.00248	1.00277	1.00147
13	2.25	0.99310	0.97441	0.94874	0.92481	0.92105	0.90536
14	2.75	1.00263	1.00272	1.00465	1.00473	1.00558	1.00597
15	3.50	1.00170	1.00273	1.00324	1.00362	1.00365	1.00414
16	5.00	0.82877	0.82011	0.81308	0.80612	0.80590	0.80183
17	7.00	0.82661	0.81999	0.81298	0.80650	0.80578	0.80155
18	11.0	0.82703	0.81956	0.81310	0.80659	0.80617	0.80179
	Total Photon Emission	**0.99675**	**0.99065**	**0.98544**	**0.98008**	**0.97945**	**0.97489**
	Total Energy Emission	**0.99921**	**0.99687**	**0.99407**	**0.99047**	**0.99001**	**0.98483**

Table 2. Comparison of nuclide activities for a single assembly with different IHM loadings and an enrichment of 5.00 wt% U235 as a function of burnup - at 100 hours after shutdown.

	Ratio=(IHM 411 kgU/IHM 379.06 kgU) for several burnups (GWD/MTU)					
Isotope	**14**	**26**	**36.5**	**46.5**	**48**	**60**
Kr83m	1.00957	1.01761	1.02382	1.02919	1.02926	1.03332
Kr85m	1.01043	1.02120	1.03024	1.03859	1.03884	1.04640
Kr85	1.00418	1.00818	1.01177	1.01509	1.01550	1.01916
Kr87	1.01451	1.02654	1.03682	1.04659	1.04736	1.05707
Kr88	1.00304	1.02682	1.03395	1.05187	1.05091	1.05742
Xe131m	0.99789	0.99615	0.99510	0.99393	0.99420	0.99282
Xe133m	1.00000	1.00069	1.00069	1.00000	0.99928	1.00000
Xe133	1.00129	1.00156	1.00196	1.00209	1.00185	1.00237
Xe135m	1.00250	1.00188	1.00189	1.00210	1.00167	1.00253
Xe135	1.00900	1.00910	1.00988	1.00996	1.00918	1.01006
Xe138	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
I131	0.99841	0.99664	0.99593	0.99524	0.99479	0.99454
I132	0.99969	0.99819	0.99791	0.99705	0.99735	0.99650
I133	1.00214	1.00263	1.00313	1.00314	1.00240	1.00365
I134	1.01102	1.01322	1.01493	1.01666	1.01614	1.01863
I135	1.00234	1.00201	1.00202	1.00202	1.00167	1.00270